

ASSESSMENTS OF WATER INGRESS ACCIDENTS IN A MODULAR HIGH-TEMPERATURE GAS-COOLED REACTOR

FISSION REACTORS

KEYWORDS: *high-temperature gas-cooled reactor, water ingress accident, simulation*

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Severe water ingress accidents in the 200-MW HTR-module were assessed to determine the safety margins of modular pebble-bed high-temperature gas-cooled reactors (HTR-module). The 200-MW HTR-module was designed by Siemens under the criteria that no active safety protection systems were necessary because of its inherent safe nature. For simulating the behavior of the HTR-module during severe water ingress accidents, a water, steam, and helium multiphase cavity model was developed and implemented in the dynamic simulator for nuclear power plants (DSNP) simulation system. Comparisons of the DSNP simulations incorporating these models with experiments and with calculations using the time-dependent neutronics and temperature dynamics code were made to validate the simulation. The analysis of the primary circuit showed that the maximum water concentration increase in the reactor core was $<0.3 \text{ kg}/(\text{m}^3\text{s})$.

The water vaporization in the steam generator and characteristics of water transport from the steam generator to the reactor core would reduce the rate of water ingress into the reactor core. The analysis of a full cavitation of the feedwater pump showed that if the secondary circuit could be depressurized, the feedwater pump would be stopped by the full cavitation. This limits the water transported from the deaerator to the steam generator. A comprehensive simulation of the HTR-module power plant showed that the water inventory in the primary circuit was limited to $\sim 3000 \text{ kg}$. The nuclear reactivity increase caused by the water ingress would lead to a fast power excursion, which would be inherently counterbalanced by negative feedback effects. The integrity of the fuel elements, because the safety-relevant temperature limit of 1600°C is not reached in any case, is not challenged.

I. INTRODUCTION

The main safety concerns of pebble-bed high-temperature gas-cooled reactors (HTGRs) with steam cycles were addressed by mitigation of three types of design basis accidents, namely: loss of coolant accidents (LOCAs), severe air ingress accidents, and severe water ingress accidents. The decay-heat removal issue after a LOCA was resolved when the HTGR design adopted the concept of modularization.¹ In the modular HTGRs (MHTGRs), the decay-heat is removed completely by means of conduction and radiation. No convection is

necessary. The air ingress accident is a relatively slow process, although the air is able to corrode the graphite reactor core. The corrosion of graphite in the reactor core will not start until natural air circulation between the environment and the reactor core is established. Thus there is a delay of several tens of hours before corrosion would actually take place.²

The above two types of accidents were addressed previously. Compared to them, the water ingress accidents are more complicated. The water ingress causes two effects, namely the corrosion of the graphite reactor core and positive reactivity insertion due to the water ingress into a submoderated reactor core. The high pressure difference between the secondary and primary circuits

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and the positive reactivity cause the accident to be rapid and severe.

According to the analyses performed by Siemens INTERATOM (Refs. 3 and 4), a one-tube double-end break in the steam generator will cause ~600 kg water ingress into the primary circuit even with a series of quick actions to actuate the reactor protection systems. Without the activation of the reactor protection systems more water would ingress into the primary circuit. The operations of the reactor protection systems include shutting down the reactor, stopping the blower, closing the isolating valves in the inlet and outlet of the steam generator, and emptying the steam generator. Although the 600 kg water ingress will not cause severe safety consequences, the mitigation systems for water ingress accidents need to be activated through valves and signals. Thus, the philosophy of inherent safety at that point may not seem to be fully compatible with those adopted in the afterheat removal, which depends on inherent safety and passive mechanisms only, not on active actions of reactor safety systems. Therefore, a thorough study for a severe water ingress accident was necessary.

The analysis of severe water ingress accidents is important for another fundamental reason. The worldwide development of a safe nuclear energy has led to the design of a new generation of nuclear power plants, which depend more on inherent and passive safety mechanisms. These efforts have obviously improved the reactor safety by reducing the core damage frequency. However, this seems not enough for the public acceptance of nuclear energy. Engineers should find a new safety philosophy to improve the public acceptance, which to a great extent is not a technical but a political and psychological issue. The so-called catastrophe-free nuclear technology adopted such philosophy.⁵ Catastrophe-free nuclear technology is realized if the radioactive fission products in all possible accidents are retained nearly totally inside the reactor plant.⁵ In a so-defined catastrophe-free nuclear plant there is no probability that to any degree the public safety is challenged. The HTR-module was meant to implement the catastrophe-free safety concept. It is under these premises that it is necessary to demonstrate that the reactor is catastrophe-free during water ingress accidents.

In order to demonstrate the inherent safety of the HTR-module under water ingress accidents, and also to determine whether the HTR-module is catastrophe-free during severe water ingress accidents, an assessment was made for the extremely severe consequences of water ingress accidents in the HTR-module by numerical simulations. All numerical analyses are performed under the general assumptions of no active actions of the reactor safety system, namely

1. no scram
2. no steam generator isolating
3. no steam generator emptying

4. no control on the feed water pump and the secondary circuit
5. no control on the blower.

In the water ingress accident, the positive reactivity insertion is a short-term effect, which is mainly determined by the rate of water ingress into the reactor core. The corrosion of the graphite reactor core is a long-term effect, which is influenced by the water ingress mass. The question of how fast the water could enter into the reactor core was addressed by simulating the accidental water transport in the primary circuit. The question of how much water could enter into the primary circuit was addressed by analyzing the behavior of the secondary circuit during water ingress accidents. Sections III and IV address these two subjects, respectively.

Besides the HTR-module,^{3,4} there were several other water ingress accident analyses for the pebble-bed HTGRs. Wawrzik established a lumped model to simulate the water ingress accident in the AVR reactor.⁶ In this model, liquid water, steam, and helium were considered. However, the lumped model used was unable to analyze the water transport from the steam generator to the reactor core. The REACT/THERMIX code⁷ could deal with the transport of steam, helium, and other gases in the primary circuit and the chemical reactions. That code was used to analyze the PNP-500 conceptual design.⁸ However, it did not consider the behavior of liquid water and the reactor nuclear power feedback. The time-dependent neutronics and temperatures (TINTE) code system⁹ combines a two-dimensional transient analysis of thermal hydraulics and nuclear dynamics with a one-dimensional simulation of the primary circuit. With these capabilities and the capability of analyzing the chemical reactions of graphite, TINTE is a more advanced analysis tool. In its current version, TINTE does not yet consider liquid water in the primary circuit. Until now there have been no computer codes available that combine the secondary circuit into a full-scale water ingress simulation. Therefore, a simulation program was developed that combined the models of liquid water vaporization and condensation in the primary circuit and the modeling of the secondary circuit in order to address the above-mentioned questions.

II. SIMULATION MODELS AND VALIDATION

II.A. Dynamic Simulator for Nuclear Power Plants System

Dynamic simulator for nuclear power plants¹⁰ (DSNP) is a block-oriented simulation language by which a large variety of nuclear power plants can be simulated. In the past the DSNP was successfully applied to LMFBR and pressurized water reactor accident analyses.^{11,12} It was also used in the simulations of pebble-bed HTGRs including the full-scale simulations of the PNP-500 plant¹³

and the THTR-300 power plant. For this reason, many new component models, e.g. the pebble-bed core, the blower, and the steam generator, were developed and extensively validated.

In the previous DSNP library there were two types of cavity models, namely a gas cavity model and a two-phase water steam cavity model. However, the water ingress from the high-pressure secondary circuit into the primary circuit is a complex process, which involves liquid water, steam, and helium multiphase flow for mass and heat transfer phenomena. There is mass transport between the liquid water and steam due to water vaporization and steam condensation. Therefore, a multiphase cavity model was developed in the DSNP system in order to have the capability to analyze the water transport in the primary circuit.

II.B. Multiphase Cavity Model

The basic assumptions include homogeneous mixture and thermal equilibrium. The sedimentation of water droplets in the cavity is not considered. For brevity, the water in liquid state was designated as “water,” water in gaseous state as “steam” and the mixture of both as “H₂O.”

Figure 1 gives the structure of multiphase cavity model. Three parameters were chosen to describe the water, steam, and helium multiphase cavity model as follows:

The helium mass fraction

$$y = \frac{M_h}{M}, \quad (1)$$

specific volume

$$v = \frac{V}{M}, \quad (2)$$

and specific internal energy

$$u = \frac{U}{M}, \quad (3)$$

where M , U , and V denote the total mass, the total internal energy, and the total volume in the cavity, respectively; and the subscript h denotes the fraction of helium.

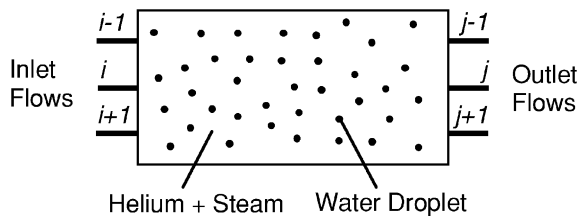


Fig. 1. Structure of a multiphase cavity.

The following conservation equations were used to determine the three variables above:

The mass conservation equation

$$V \frac{d\left(\frac{1}{v}\right)}{dt} = \sum_i G_i^{in} - \sum_j G_j^{ex}, \quad (4)$$

the mass conservation equation of helium

$$M \frac{dy}{dt} = \sum_i y_i^{in} G_i^{in} - y \sum_j G_j^{ex} - y \left(\sum_i G_i^{in} - \sum_j G_j^{ex} \right), \quad (5)$$

and the energy conservation equation

$$M \frac{du}{dt} = Q + \sum_i h_i^{in} G_i^{in} - h \sum_j G_j^{ex} - u \left(\sum_i G_i^{in} - \sum_j G_j^{ex} \right) + K(T_m - T), \quad (6)$$

where t , T , G , h , and Q denote time, temperature, mass flow rate, specific enthalpy, and heat source term respectively; K is the product of the heat transfer coefficient and the wall surface area; the subscript m denotes the cavity wall; and the superscripts in and ex denote the inlet and outlet of the cavity.

Then the specific volume and the specific internal energy of H₂O can be written as

$$v_{H_2O} = \frac{v}{1-y} \quad (7)$$

and

$$u_{H_2O} = \frac{u - yC_v T}{1-y}, \quad (8)$$

where C_v is the specific heat capacity of helium at constant volume.

Using (v_{H_2O}, u_{H_2O}) as input variables, the temperature can be determined by iterations and the other properties like steam partial pressure and steam quality, etc., can be calculated through a water thermodynamic calculation.

The specific volume of helium is

$$v_h = \frac{v}{y} \left[1 - \frac{(1-x)v_w}{v_{H_2O}} \right], \quad (9)$$

where x is steam quality and the subscript w denotes liquid water. The helium partial pressure is

$$p_h = \frac{RT}{\mu_h v_h}; \quad (10)$$

the pressure of cavity is therefore

$$p = p_h + p_{\text{H}_2\text{O}}, \quad (11)$$

where R is the general gas constant, and μ is molar mass.

On the basis of the previous DSNP two-phase model, the multiphase cavity model is formed by adding a differential equation on the helium fraction y as in Eq. (5) and implemented in the DSNP system as a component model. A material property subroutine that determines the thermodynamic properties of the cavity by using (y, v, u) as input has been implemented as a user-supplied library, which can also be used in other DSNP components easily. In addition, the hydraulic network in DSNP has also been improved to deal with the multiphase flow.

II.C. TINTÉ Program

With the one-dimensional DSNP simulation, it is impossible to obtain the fuel peak temperature during the transient, which is an important safety parameter to assess the consequences of severe water ingress accidents. TINTÉ is a program with coupled two-dimensional core neutron dynamics and thermal hydraulics in HTGRs (Ref. 9). The two-dimensional TINTÉ simulation, based on the results given by the DSNP simulation, can give the detailed fuel temperature profiles. With the updated simulation capability, it became feasible to analyze the consequence of reactivity insertion caused by the water ingress.

II.D. Validation of Models

The Marviken III Test¹⁴ is one in a series of tests performed as a multinational project at the Marviken Power Station by ABB Atom Sweden. In order to validate the DSNP model, the Marviken III Test 4, which aimed to establish the choked flow rate data for a large-scale nozzle with subcooled and low-quality water conditions at the nozzle inlet, was simulated. As shown in Fig. 2, the pressure vessel was initially filled with water

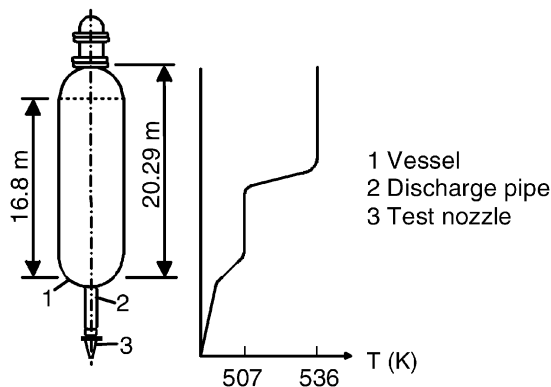


Fig. 2. Marviken III Test.

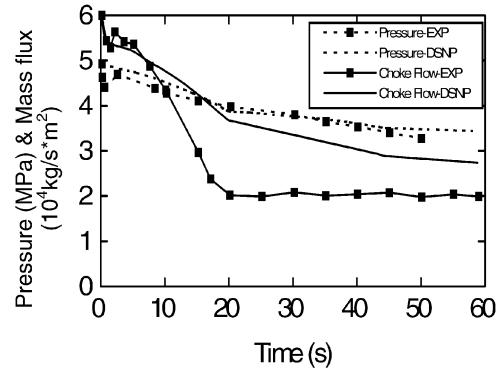


Fig. 3. Comparison of DSNP with Marviken test.

to an elevation of 16.8 m above the discharge pipe inlet. The steam dome above the water was saturated at 4.94 MPa. The water of ~ 6 m in depth below the surface was at nearly saturation conditions. The water below the saturated fluid and a small transition zone was subcooled by $\sim 30^\circ\text{C}$. An experiment was performed by opening the discharge pipe outlet to the ambient pressure.

As shown in Fig. 3, the DSNP simulation gives a comparable pressure history at the top of the experiment vessel and a comparable choked flow rate. However, because of lacking a nonequilibrium model, the code was unable to predict the pressure undershooting at the beginning of the blowdown. This comparison shows that the DSNP could give a not-exact but reasonable prediction for a large-scale discharge test.

In order to validate the multiphase cavity model, a process described as below was simulated. Subcooled water at conditions of 19 MPa/ 240°C enters into a cavity of helium at conditions of 6 MPa/ 250°C . The flow rate of water is 0.1 kg/s. The cavity volume was estimated as 1.355 m³. The calculation gave reasonable results as shown in Fig. 4. After entering the cavity, water became vaporized due to a very low partial pressure of steam. Because

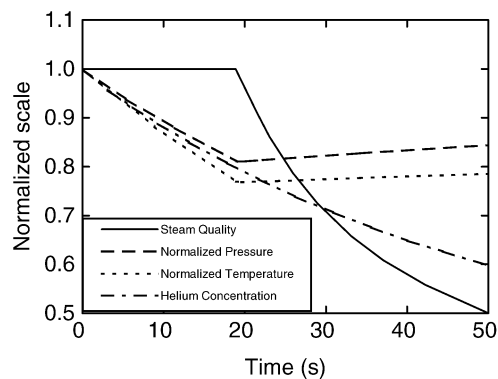


Fig. 4. Results of helium cavity with subcooled water ingress.

the subcooled water can not compensate the energy required in the vaporization, the water has to receive the energy from the helium. Therefore, the temperature and then the partial pressure of helium were decreased. During the early phase, the water in the cavity was vaporized completely to superheated steam. Afterwards, since the helium was unable to compensate the vaporization energy, liquid droplets or mist were formed in the cavity. The system pressure and temperature reached their minimum point and increased thereafter.

For comparing the DSNP model with the TINTE program, the blower shutdown transient in the HTR-module was simulated. The transient was initiated from a normal operation. No control rods were moved during the transient. As the blower speed was reduced, the flow rate of the primary circuit was decreased, and the helium temperature at the reactor core outlet was increased. The increase of coolant temperature and fuel temperature would lower the reactor nuclear power through the negative temperature reactivity feedback. The helium flow rates and the reactor nuclear power during the transient from both calculations were in good agreement. This indicates that the blower characteristics and the hydraulic calculation based on DSNP are very close to the TINTE results. In addition, both TINTE and DSNP modeling could predict the natural circulation in the primary circuit after the blower was totally stopped. The difference between both calculations was within 6%.

III. ASSESSMENT OF WATER INGRESS RATE

For a low water ingress rate, the slow positive reactivity insertion can be rapidly compensated by the negative fuel temperature reactivity feedback. For a large water ingress rate, the negative temperature feedback is unable to compensate the positive reactivity quickly, and the core peak power would increase. This would probably make the fuel peak temperature exceed the safety-relevant limit in such cases. Therefore, it is important and necessary to study the effect of water ingress rate on the consequence of accidents.

III.A. General Characteristics of Water Ingress Accidents

Figure 5 gives an overview of the HTR-module.¹⁵ A break in the steam generator initiates the water ingress accident. Before the H₂O reaches the reactor core, it travels through the blower, the upper space of the steam generator, the connecting vessel, the reactor bottom space, and the coolant holes in the side reflector as shown in Fig. 5. Such a long mass transport path will delay the H₂O ingress into the reactor core.

The centrifugal separation of the water droplets in the blower is a dominant effect for the water transport in the blower, which has a major impact on the behavior of

water transport to the reactor core. Steam is impossible to separate from the mixture; therefore, the influence of blower separation will depend on the amount of the water droplets that flow through the blower.

The break positions in the steam generator will greatly influence the H₂O transport process in the primary circuit. According to the thermodynamic properties of the steam generator in the HTR-module shown in Table I, it is conceivable that a break in the subcooled section of the steam generator will lead to the flashing of water in the primary circuit by receiving energy from the helium and the wall. Due to the delay of heat release from the wall, the instantaneous effect of the subcooled water ingress could cause a temperature reduction near the break, and therefore possibly a pressure reduction in the primary circuit. The continuous water ingress will cause the H₂O in the primary circuit to reach a saturated state as the partial pressure increases and temperature decreases. The temperature in the primary circuit will then no longer decrease. Small water droplets in mist form will appear due to the condensation of the steam. The large amounts of stored heat in the wall could afterwards cause the temperature to increase and cause the water droplets to evaporate.

If the break occurs in the superheated section of the steam generator, it will give a quite different scenario. The superheated steam will cause the temperature and pressure in the primary circuit to increase and remain in the steam state during transport to the reactor core. Only under a very extreme condition, i.e., when the H₂O partial pressure reaches 4 MPa, will the steam at 250°C become saturated. Thus, the appearance of water droplets is very unlikely.

TABLE I

Brief Parameters and Design Data of the Steam Generator

	Primary	Secondary
Parameter		
Thermal power (MW)	202	202
Flow rate (kg/s)	85	77
Inlet temperature (°C)	700	170
Outlet temperature (°C)	250	530
Inlet pressure (MPa)	6	21
Pressure drop (MPa)	0.04	2
Design data		
Heat transfer area (m ²)		2100
Number of heat transfer tubes		220
Dimension of heat transfer tube (mm)		23 × 4.2/23 × 2.5
Dimension of compensate tube (mm)		21.1 × 3.3
Height of heat transfer tube (mm)		8200
Material of heat transfer tube		Incoloy 800

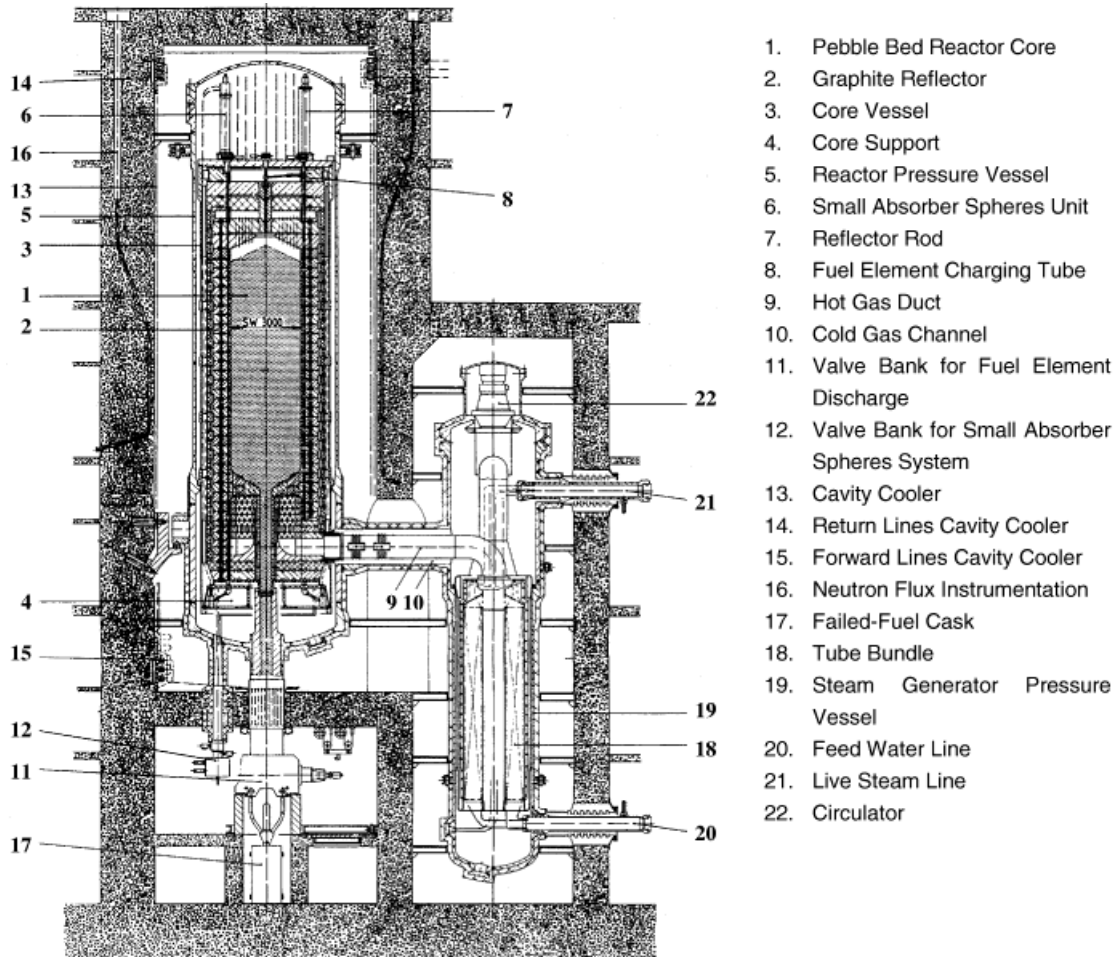


Fig. 5. Overview of HTR-module primary circuit.

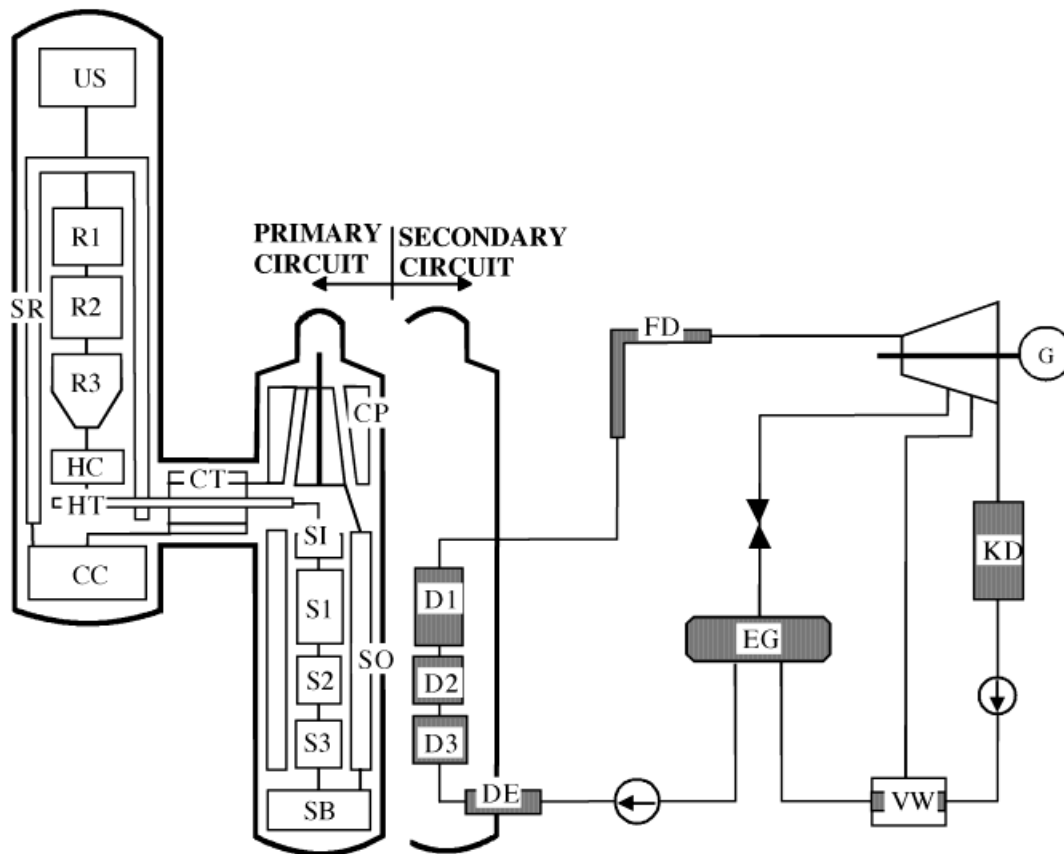
The spectrum of the steam generator break sizes could range from a small crack in a heat transfer tube to double-end breaks of one tube, several tubes, or even all 220 tubes. The extreme condition is a double-end break of the $\phi 550$ mm plenum at the inlet and outlet of the steam generator. A spectrum of breaks were considered in order to find the most limiting case for the severe water ingress accidents.

III.B. Accidental Water Transport in the Primary Circuit

The flow chart of the DSNP modeling of the HTR-module primary circuit can be found in Fig. 6. The fuel thermal response is simulated by a lumped wall energy conservation equation. The reactor power is considered as a heat source to the wall. The wall heat storage is therefore considered, and the heat removal to the secondary circuit is referred to as a heat sink for the wall of the steam generator. The total gas volume in the primary circuit is 237 m^3 . The nonflow gas volume of $\sim 120 \text{ m}^3$ at the reactor top and at the gap between core vessel and

the reactor pressure vessel is excluded in the current modeling because it would not affect the accidental water transport in the primary circuit. In addition, the nuclear feedback is not included. The reactor power was kept at 200 MW, and the heat was removed from the steam generator at the same value. This was to isolate the study on the pure thermal-hydraulic phenomena of the accidental water transport, which would not affect the analysis results of water ingress rate.

Subcooled water was assumed and was injected at the bottom of the steam generator with a constant flow rate of 100 kg/s . After 2000 kg of water was injected, the water flow rate was taken as zero. As shown in Fig. 7, the H_2O concentration, i.e. the mass fraction of H_2O in the $\text{H}_2\text{O}/\text{helium}$ mixture, rises quite quickly in the steam generator from 0 to 0.55 in 2 s. However, the H_2O concentration in the reactor core increases rather slowly because of the long transport path. The reactor core became affected by the appearance of H_2O in $\sim 3 \text{ s}$. The delay time, which depends on the blower speed and water ingress flow rate, is just the time necessary for H_2O to



US – Upper non-flow space; R1,R2,R3 – Reactor core volumes; SR – Coolant channel in the side reflector; HC – Hot gas cavity in the bottom reflector; HT,CT – Hot and cold gas tubes in the connecting vessel; CC – Cold gas cavity at the bottom of reactor vessel; CP – Cold gas plenum at the outlet of blower; SI – Steam generator inlet; S1,S2,S3 – Steam generator boundaries; SB – Steam generator bottom; SO – Steam generator outlet; D1,D2,D3 – Steam generator superheated, two-phase and subcooled sections; DE – Steam generator inlet; FD – High pressure steam pipe; KD – condenser; VW – Preheater; EG – Deaerator.

Fig. 6. DSNP modeling of the HTR-module power plant.

travel from the steam generator to the core inlet. The rate of H_2O concentration increase in the reactor core, from 0 to 0.55 in 20 s, is about 10 times smaller than that in the steam generator. This phenomenon is important to the assessment of safety features because a lower increasing rate of the H_2O concentration will reduce the reactor power peak.

III.C. Rate Limit of Water Ingress

In order to find the limit on the rate of water ingress progressing into the reactor core, different water ingress flow rates of 10, 100, 1000, and 10000 kg/s were proposed, even though not every proposed value was practical. The total water ingress mass in these cases was assumed the same to all cases, namely 2000 kg. As shown

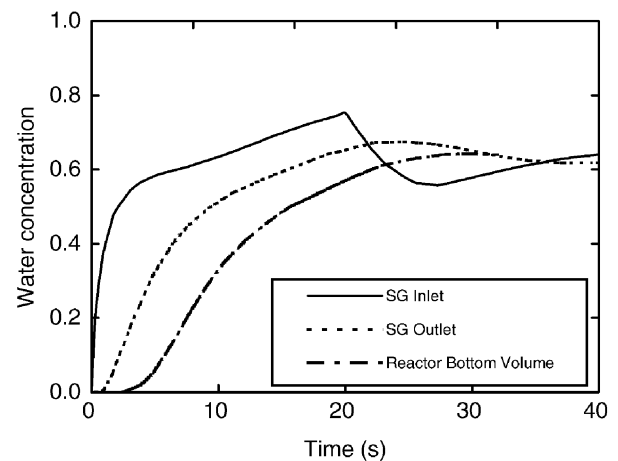


Fig. 7. Water concentration in different components.

in Fig. 8, the H_2O density increasing rate for the case of 100 kg/s is 5.8 kg/m³ in 30 s. The values for the cases of 1000 and 10000 kg/s are nearly the same, namely 5.8 kg/m³ in ~20 s. It means that the H_2O density increase rate reaches a limit, i.e., 0.3 kg/(m³s). This was due to two mechanisms: (1) the transport of H_2O is limited by the system geometry, and (2) the pressure decrease at the beginning due to the water vaporization behaves like a spring, which reduces the H_2O impulse in the reactor core.

The heat release from the wall plays an important role on the accidental water transport in primary circuit. In order to examine the influence of wall mass, cases of the wall mass of 50% to 150% of the original one were considered. The calculated results of these cases show that, for different wall masses, there are no major changes in the rate of water ingress progressing into the reactor core. However, the small wall mass seems to be unable to evaporate all water droplets, and it is possible that the water droplets would enter the reactor core at a later time.

In order to determine the blower separation effect liveness, the separation efficiency was defined as the ratio of the water mass separated over the total water mass passing the blower. The separation efficiency depends on the diameter of the water droplets, the blower speed, and the geometry of the blower. To obtain the efficiency, it was necessary to perform sensitivity studies on the blower. The separation efficiencies 0%, 25%, 50%, and 100% were assumed. Figure 9 shows that the 100% separation of water droplets in the blower reduces the H_2O density in the reactor core to half of the value of the no-separation case. The increasing rate of the H_2O density is also lowered. Therefore, the blower separation effect is a positive factor for mitigating the consequences of severe water ingress accidents.

The previous analyses were based on a break in the subcooled section of the steam generator. A break in the superheated section of the steam generator has a higher probability for its occurrence during the operation of the

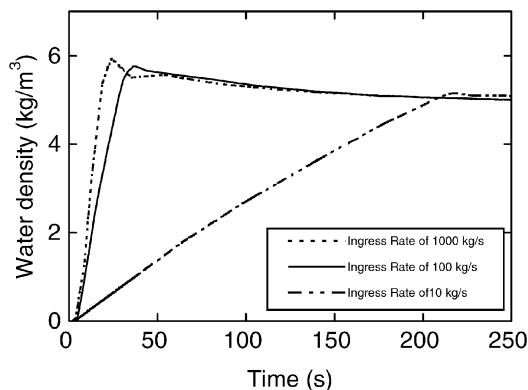


Fig. 8. Average water density in core for different water ingress flow rates.

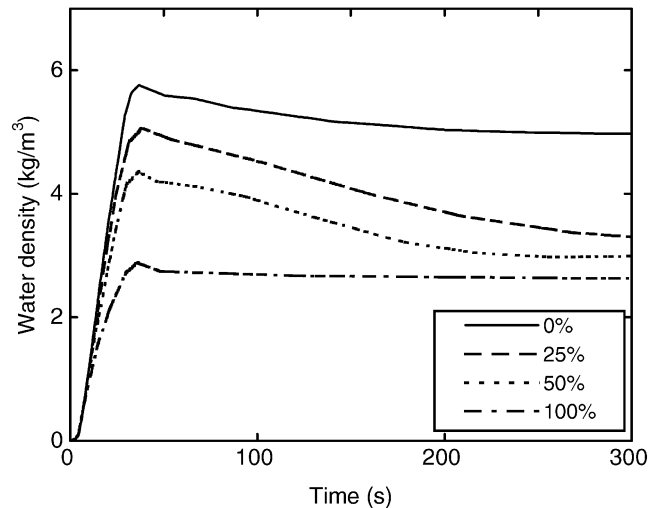


Fig. 9. Average water density in core for different separation efficiencies.

HTR-module power plant because of the higher operating temperature. The steam was assumed to be injected from the superheated section of the steam generator to the primary circuit with a flow rate of 10 kg/s. The total steam ingress mass was 2000 kg. The steam flow rate of 10 kg/s was equal to double-end breaks of about 11 heating tubes of the steam generator, which was very unlikely from the engineering viewpoint. Because there was no condensation or vaporization of H_2O in the steam ingress process, the temperatures in the primary circuit did not change very quickly. However, the system pressure increased quickly. At 100 s, the pressure approaches 9 MPa, which was far higher than the opening pressure of the safety relief valves. In the analyses presented in this section, the valves were assumed closed. It seems that a break in the superheated section is not so much more detrimental than that in the subcooled section. The current paper therefore focuses on the break in the subcooled section.

From all analyses presented above, the maximum rate of H_2O ingress into the reactor core was determined to be 0.3 kg/(m³s).

III.D. Maximum Peak Fuel Temperature

To find the maximum peak fuel temperature, the TINTE program was used to perform a two-dimensional calculation. The simulation was performed by artificially injecting the steam from the annular channel of the coaxial tube. The system pressure was kept at 6 MPa, the rated value.

Figure 10 gives the core maximum temperature versus the helium flow rate and the water flow rate. The most limiting situation is the 200 kg/s steam ingress flow

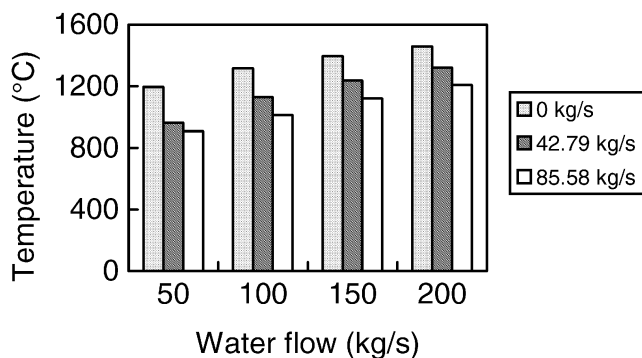


Fig. 10. Fuel element maximum temperatures for the different water ingress conditions.

rate with 0 kg/s helium flow rate. As the steam flows into the reactor core, the positive reactivity insertion quickly caused the reactor power to increase, which reaches the peak value of 28 times of the rated value rapidly. Thereafter, the power is reduced by the negative temperature reactivity feedback. The sharp power increase heats the core quickly. In a short time of 7 to 8 s, the core peak temperature increases from 823 to 1458°C before it decreases due to the power decreasing. At ~15 s, the steam replaces all the helium in the core. After that, because of the pressure in the reactor, the water inventory in the reactor maintains approximately constant. No further positive reactivity is inserted. The core maximum temperature is lower than the safety-relevant temperature limit of 1600°C.

The 200 kg/s steam input is equal to a H₂O density-increasing rate of 5 kg/(m³s). It is approximately 16 times of the limiting value estimated by the above analyses, namely 0.3 kg/(m³s). The analyses in this section seem to indicate that no significant core damage would occur during the primary water ingress power impulse under any circumstances.

IV. ASSESSMENT OF WATER INGRESS MASS

In the secondary circuit of the HTR-module (see Fig. 6), the deaerator contains ~200 000 kg of water. If such a huge amount of water was pumped into the primary circuit continuously for a long time, the reactor core could be filled with water. This would cause severe chemical reactions between the water and reactor graphite fuel elements. In the HTR-module design, there are many operating options to limit the amount of water moving into the primary circuit during a water ingress accident. However, if the safety protection system fails, is it possible that nearly all water in the deaerator is pumped to the steam generator and then to the primary circuit? In this section, this question is addressed by analyzing the

behavior of the secondary circuit during the severe water ingress accidents and by performing the full-scale simulation of severe water ingress accidents.

IV.A. Phenomena of Cavitation in the Feedwater Pumps

Pumps are normally designed to function with liquid as the working fluids. To make a pump work properly, the working fluids in the pump should be kept in subcooled states to avoid bubbles appearing on the blade surface. This is because bubbles would form cavities on the blade surface, and if the bubbles last for a long time during operation, the pump would fail. This is called cavitation. An extreme situation is that the main flow stream is saturated, what is referred to as full cavitation. In this case, the pump head will quickly decrease following a strong vibration.

For the HTR-module, the secondary circuit is so designed that during normal and transient operations no cavitation is possible in the feedwater pump. However, during the severe water ingress accidents, the pump may behave differently. A break in the steam generator will depressurize the secondary side of the steam generator. Consequently, the pump head is lowered. According to the general characteristics of pumps, as the pump head decreases the flow rate in the pump will increase. The flow rate increase will reduce the system available net positive suction head (NPSH) and raise the system required NPSH. If the former is less than the latter, the pump cavitates and the pumping rate is reduced. Thus, it is possible a full cavitation could occur if there is enough depressurization.

For a typical condensate pump in a power plant, the full cavitation starts after the system flow rate exceeds 115%. However, the corresponding data for the HTR-module is unavailable because no HTR-module has ever been built. In the current study, it was conservatively assumed that the pump will be in full cavitation if the pump flow rate exceeds the rated value by 30%.

IV.B. Behavior of the Secondary Circuit During Water Ingress Accidents

The DSNP model of the secondary circuit can be found in Fig. 6. The pump characteristics are assumed to be

$$H = 3614 - 1319(G/77)^{1.8}, \quad (12)$$

where H is the pump head (m) and G is the pump flow rate (kg/s). It is close to the characteristic of the feed-water pump in the THTR-300.

The water discharge flow rates at the steam generator bottom were assumed as 10, 50, and 100 kg/s. The calculation was performed with the assumption that there was no safety protection. The results are shown in Fig. 11. The 10 kg/s discharge flow rate does not lead the pump to a full cavitation state. The system pressure of the

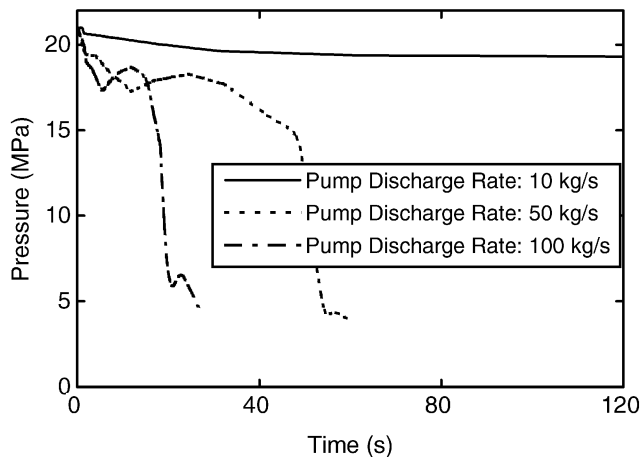


Fig. 11. System pressure of the secondary circuit with considering pump cavitation.

secondary circuit is stabilized at 19.5 MPa. The discharge flow rates of 50 and 100 kg/s lead the pump to a full cavitation state. The pressure of the secondary circuit drops sharply to the equilibrium pressure of the primary circuit.

The results demonstrate that during a severe water ingress accident, the cavitation will quickly stop the feedwater pump and limit the water ingress to the primary circuit. This is an inherent safety feature. It seems that a continuous large flow rate of the water ingress from the steam generator is impossible. However, the results also indicate that the cavitation will come into effect only for a large break, such as multitube breaks. A small break may not be enough to lead to a full cavitation state. In this case the secondary circuit will regulate itself to a new steady state and the water will be continuously pumped to the primary circuit. Thus, the long term water ingress into the reactor primary circuit will be limited to a value between 10 and 50 kg/s.

IV.C. Full-Scale Simulation of Severe Water Ingress Accidents

The model of full-scale simulation of the HTR-module power plant is illustrated in Fig. 6. The pressures of both circuits are important for the full-scale simulation, thus the nonflow volume that simulates the space between the core vessel and the reactor pressure vessel has been considered important as it improves the modeling of the pressure transient. The water ingress reactivity insertion, as a function of core average H_2O density, has been included in the nuclear feedback model.

There are two safety relief valves in the primary circuit. The first opens when the pressure exceeds 6.9 MPa with a cross section area of 0.5 cm². The second opens when the pressure exceeds 7.3 MPa with a cross

section area of 33 cm². The safety relief valves are assumed to fail to close during the accidents.

Figure 12 summarizes the time-dependent pressures of the primary and secondary circuits for the water ingress accidents with one-tube double-end breaks. For the break in the subcooled water section of the steam generator, i.e., water section break, the pressure of the primary circuit increases gradually. At the time of 21 s, the pressure of the primary circuit reaches 6.9 MPa. Soon after that the first safety relief valve opens. The break flow rate at this time is 56.8 kg/s. The discharge flow rate of the safety relief valve is about 0.3 kg/s, which is not large enough to depressurize the primary circuit. The pressure of the primary circuit increases continuously until ~27 s, at which time the second safety relief valve opens. About 24 kg/s of the H_2O /helium mixture is discharged to the reactor cavity and causes the primary circuit to depressurize. When the secondary circuit is depressurized to ~9 MPa, the full cavitation of the feedwater pump occurs based on the assumption that the 130% flow rate in the feedwater pump was the threshold point of full cavitation. The secondary circuit is depressurized quickly due to loss of driving force and the water ingress stops. The maximum H_2O inventory is limited within 1700 kg due to the full cavitation of the feedwater pump, as shown in Fig. 13.

For the one-tube break in the superheated steam section of the steam generator, i.e., steam section break, the mass inventory curve is smoother than that of a water section break. At the time of ~140 s, the first safety relief valve opens and the primary circuit is depressurized. The secondary circuit pressure at the steam generator bottom will be stabilized at ~18 MPa. At this pressure, there will be no cavitation in the feedwater pump. However, the maximum H_2O inventory is plateaued to a value slightly below 1200 kg due to the opening of the safety relief valve.

For the water section 10-tube break, due to the sharp depressurization of the secondary circuit, the pressure

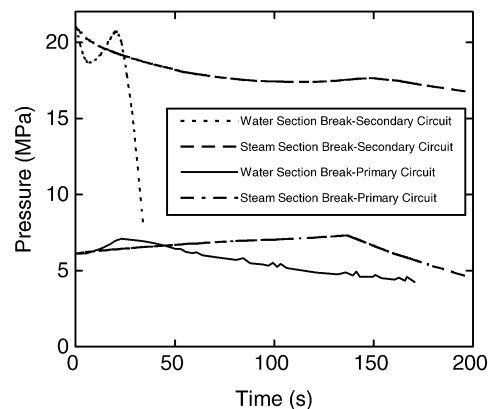


Fig. 12. Pressures of the primary and secondary circuits for the one-tube double-end break water ingress accidents.

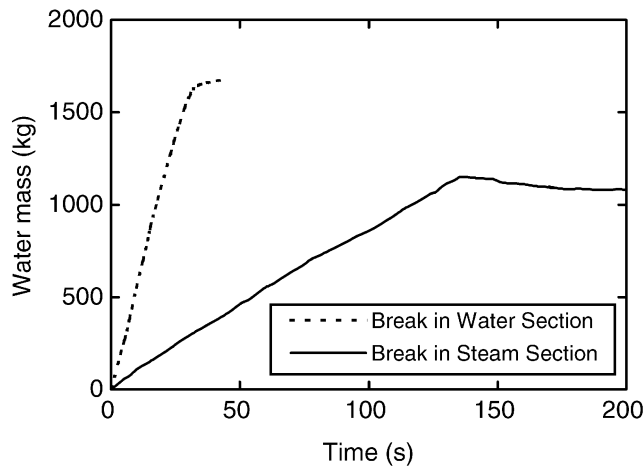


Fig. 13. H₂O inventory in the primary circuit for the one-tube double-end break water ingress accidents.

equilibrium between the primary and secondary circuit is reached within 10 s. Thereafter, the pressures of the primary and secondary circuits move together because no water is pumped from the deaerator. Because a full cavitation of the feedwater pump is induced very quickly, the maximum H₂O inventory in the primary circuit is limited to 1600 kg, as illustrated in Fig. 14. Passing the peak of the curve, the H₂O inventory drops down because of discharge in the safety relief valve.

For a 10-tube break in the steam section, the pressure decrease in the secondary circuit is rather smooth, and the full cavitation of the feedwater pump can not be induced in a short time. At the time of 80 s, the H₂O inventory in the primary circuit is ~2600 kg. Because the discharge flow rate of the safety relief valves is not large enough, the H₂O inventory will continually increase until a full cavitation of the feedwater pump occurs. If a larger cross section area of the safety relief

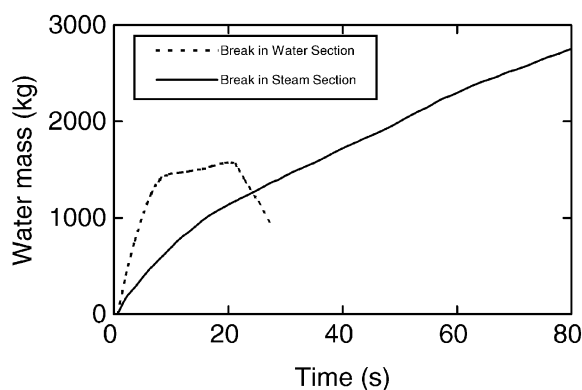


Fig. 14. H₂O inventory in the primary circuit for the 10-tube double-end break water ingress accidents.

valve could be designed, the H₂O inventory could be limited to 3000 kg.

For a plenum break in the steam section, the depressurization of the secondary circuit at the beginning is very fast. The feedwater pump will go full cavitation in ~15 s. For the feedwater section plenum break, due to a sharp depressurization of the secondary circuit, the feedwater pump could not function at all. In any case the feedwater pump will be stopped because of a full cavitation; in this case, only the 2830 kg of H₂O retained in the steam generator could flow into the primary circuit. Therefore, the total mass of H₂O ingress into the primary circuit is no more than 2000 to 3000 kg.

IV.D. Limitation of Water Ingress Accidents

From the analyses above, it seems that the total H₂O inventory in the primary circuit can be assumed to be limited within 2000 to 3000 kg for all possible break sizes by the following arguments:

1. for a small-size break ($\ll 1$ -tube break), the water ingress flow rate is so small that there is enough time to let the operator take correct actions and the accident can be resolved by the operation and control of the helium system.
2. for a middle-size break (< 1 -tube break), the depressurization of the secondary circuit is not large enough to cause the full cavitation of the feedwater pump. The opening of the safety relief valves in the primary circuit will limit the H₂O inventory in the primary circuit.
3. for a large-size break (\sim or > 1 -tube break), the depressurization of the secondary circuit is large enough to cause the full cavitation of feedwater pump. The stop of the feedwater pump and the discharge by the safety relief valves limit the H₂O inventory in the primary circuit.

The boundary between the mid-size break and the large-size break is determined by the characteristics of the feedwater pump. A break that is large enough to induce the full cavitation of the feedwater pump is defined as a large-size break. The size of the safety relief valve should be designed in a way that its opening can limit the H₂O inventory in the primary circuit.

In this context, the reactor safety does not depend on the proactive actuation of the safety protection system such as the steam generator isolating and emptying. It depends only on the inherent or passive mechanisms; such as

1. the opening of the safety relief valve
2. the full cavitation of the feed water pump.

During water ingress accidents, in order to control chemical reactions between H₂O and fuel elements on a long-term basis, the most effective method is to shut down the reactor. After the reactor is shut down, the

temperature in the core will soon become $<500^{\circ}\text{C}$. At this temperature the corrosion of the fuel elements by the H_2O is minimum. The reactor shut down by control rods is adopted only for the purpose of minimizing the long-term reactivity effects. There is no need for a quick actuation of a shut down reactor system.

V. CONCLUSIONS

For the HTR-module, during the severe water ingress accidents without any proactive actuation of safety protection systems, the liquid water vaporization in the steam generator and characteristics of H_2O transport from the steam generator to the reactor core reduce the rate of water ingress into the reactor core. The maximum H_2O concentration ingress rate into the reactor core was determined as $0.3 \text{ kg}/(\text{m}^3\text{s})$. If the secondary circuit is depressurized enough due to a larger break in the steam generator, the feedwater pump will be inherently stopped by a full cavitation. The total H_2O inventory in the primary circuit can be limited within 3000 kg for all possible break sizes. Although severe water ingress accidents involve a significant increase of reactivity, the impact of the fast nuclear transient on the fuel element integrity does not lead to any safety concerns. Especially there is no indication of release of larger amounts of radioactivity from the fuel elements under any circumstances.

The assessments made in this paper are yet to be verified experimentally. These include the delay effects in the primary circuit and the behavior of the feedwater pump. Modeling of the chemical reactions is not included in the current scope. This will be improved in the future. The thermodynamic nonequilibrium could be a dominant effect during fast water ingress accidents. The detailed analysis with such, which although not expected to alter the current conclusions, would be nevertheless included in the future investigation.

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